June 27, 2003

Mr. Bryce Shriver
Senior Vice President and
Chief Nuclear Officer
Susquehanna Steam Electric Station
PPL Susquehanna, LLC
769 Salem Blvd., NUCSB3
Berwick, PA 18603-0035

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION
NRC INSPECTION REPORT 50-387/03-006 & 50-388/03-006

Dear Mr. Shriver:

On May 16, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed a team inspection at the Susquehanna Steam Electric Station, Units 1 and 2. The enclosed report presents the results of that inspection, which were discussed with you and members of your staff on May 16, 2003.

The inspection examined activities conducted under your license as they relate to the design and performance capability of the residual heat removal and the emergency diesel generator systems, and compliance with the Commission's rules and regulations and with the conditions of your license. The inspection consisted of system walkdowns; examination of selected procedures, drawings, modifications, calculations, surveillance tests and maintenance records; and interviews with site personnel.

Based on the results of this inspection, the team identified two findings of very low safety significance (Green), both of which were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these two findings as non-cited violations (NCVs), consistent with Section VI.A of the NRC's Enforcement Policy. If you contest these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Susquehanna Steam Electric Station.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Lawrence T. Doerflein, Chief
Systems Branch
Division of Reactor Safety

Docket Nos. 50-387, 50-388
License No. NPF-14, NPF-22

Enclosure: Inspection Report 50-387/03-006 & 50-388/03-006
w/Attachment: Supplemental Information

cc w/encl:
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REGION I

Docket Nos.	50-387, 50-388
License Nos.	NPF-14, NPF-22
Report Nos.	50-387/03-006, 50-388/03-006
Licensee:	PPL Susquehanna, LLC
Facility:	Susquehanna Steam Electric Station
Location:	Berwick, PA 18603-0035
Dates:	April 28-May 16, 2003
Inspectors:	H. Gray, Senior Reactor Inspector, DRS, Team Leader B. Bickett, Reactor Inspector, DRS F. Arner, Senior Project Engineer, DRP S. Chaudhary, Senior Reactor Engineer, DRS G. Morris, Reactor Inspector, DRS V. Ruuska, Inspection Branch Chief, Stuk, Finland S. Spiegelman, NRC contractor A. Ziedonis, Engineering Co-op student
Approved By:	Lawrence T. Doerflein, Chief Systems Branch Division of Reactor Safety

Summary of Findings

IR 05000387/2003-006 and 05000388/2003-006 ; on 04/28/03 - 05/16/03, Susquehanna Steam Electric Station, Units 1 and 2; Safety System Design and Performance Capability.

The inspection was conducted by five region-based inspectors, an inspection supervisor from Stuk (Finland), a co-op engineering student and one NRC contractor. Two findings of very low safety significance (Green) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be “Green” or may be assigned another severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000**.**

1. Inspector Identified Findings

Cornerstone: Mitigating Systems and Barrier Integrity

- Green. A non-cited violation of 10 CFR 50, Appendix B, Criterion V, was identified regarding inadequate procedural guidance for placing the residual heat removal system (RHR) suppression pool cooling in service during a condition of low RHR loop pressure.

  The finding was determined to be greater than minor because it affected the mitigating systems and barrier integrity objectives of the suppression pool cooling (SPC) function. The procedural method could have challenged the integrity of the affected RHR loop components by creating the potential for a significant water hammer condition. The finding was determined to be of very low safety significance through a SDP, Phase 3 analysis because only one train of RHR was in suppression pool cooling for a limited time period over a year, and the remaining train would be unaffected. (Section 1R21.b.1)

- Green. A non-cited violation of 10 CFR 50, Appendix B, Criterion V, was identified regarding inadequate procedural guidance for operation of RHR in the suppression pool cooling (SPC) mode with a low pressure coolant injection (LPCI) signal present.

  The finding was determined to be greater than minor because it affected the mitigating systems and barrier integrity objectives of the suppression pool cooling function, in that the hard card, a procedure attachment that summarizes the detailed steps of the procedure, associated with the SPC procedure contained steps which would have resulted in an incorrect valve alignment resulting in no flow through the RHR heat exchangers. The finding was determined to be of very low safety significance through a SDP, Phase 3 analysis because the operating procedure was correct and the operators had extensive training and practice at SPC operation. (Section 1R21.b.2)

Report Details

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Safety System Design and Performance Capability (IP 71111.21)

a. Inspection Scope

The team reviewed the Susquehanna Units 1 and 2 design and performance capability of the residual heat removal (RHR) and emergency diesel generator (EDG) systems, as well as selected interfacing and supporting systems. Using risk insights, the team focused inspection activities on components and procedures that would mitigate the effects of postulated accident sequences.

Regarding the residual heat removal system, the NRC inspection team review focused on the Low Pressure Coolant Injection (LPCI), Containment spray, and Suppression Pool Cooling (SPC) modes of operation. The inspection team reviewed the design basis document (DBD), the Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TS), design calculations, and other supporting documents to ensure the systems can be relied upon to meet their functional requirements. The team also reviewed the applicable procedures which would be used during a plant cooldown following a Loss-of-Offsite Power (LOOP) event. The team reviewed the capability of the RHR system to respond to both a transient situation (LOOP) as well as automatic LPCI signals for the Emergency Core Cooling mode.

The team reviewed the procedures used to operate and test the RHR system during both normal and accident conditions. The types of procedures reviewed included: system operating procedures, abnormal and emergency operating procedures, and surveillance tests. The team also reviewed the licensed operator training lesson plans for the RHR system to ensure they accurately described the design features of the system and were consistent with the design bases.

The operational readiness, configuration control and material conditions of the RHR system were addressed through several plant walk downs including the system loop compartments and the control room. During the walkdowns, the team verified conformance to the design drawings of the installed piping, supports, and valves. Also, the team reviewed technical rationale and evaluation for elimination of RHR pump room coolers, and the resultant change in pump room environment, and environmental qualification of other component in the pump rooms. The team interviewed key PPL personnel to ascertain performance trending methods and action plans for selected system issues.

The team reviewed selected design changes for valve modifications and strainer modifications. For the portions of the RHR systems inspected, the team reviewed the applicable Technical Specifications, UFSAR, Susquehanna licensing submittals and associated NRC safety evaluations, calculations, engineering evaluations, Susquehanna design basis documents, plant modification packages, piping and instrumentation drawings (P&IDs), electrical schematics, instrumentation and control drawings, logic diagrams, and instrument setpoint documentation. Components selected for detailed review included the RHR pumps, and the heat exchanger bypass (F048A,B) and test return valves (F024A,B).

The RHR system flow, temperature and pressure capability were reviewed for Low Pressure Coolant Injection, Containment Spray, and Suppression Pool Cooling . The Process diagram FF124510 for RHR system flow requirements was compared to calculations and UFSAR values for consistency. The Net Positive Suction Head (NPSH) for the RHR pump was reviewed for the two limiting cases of pump runout flow and long-term cooling flow at maximum suppression pool temperature. Vortex analysis for RHR pumps was evaluated against the required water level in the suppression pool to verify the minimum water level to assure the RHR pump performance would not be degraded by excessive air entrainment due to vortex formation.

The team reviewed the design changes required for the RHR system related to potential common mode failure of gate valves. The modification and analysis for bonnet cavity thermal binding of valves F015A and B were reviewed. The team also reviewed selected parts of the design changes performed for removal of the function of equipment for the steam condensing mode. In addition, the containment isolation function of valves HV151, FO50 A&B, and test requirements and results, especially of valves FO016A, FO021, FO015 were reviewed for appropriateness, technical validity, acceptability of test results, data analysis and evaluation. Valve test and surveillance data were reviewed for history of functional readiness, system availability and operability

The team also reviewed the electrical design of the diesel generators, the 125 volt batteries, the RHR and emergency service water (ESW) pump motors, and selected valves. The team reviewed design calculations, drawings, maintenance and surveillance test procedures and test results associated with those components. Included in this review was associated component control logic, electrical protection device and instrumentation adequacies. The team also performed walkdowns of the diesels generators and batteries, and observed diesel generator testing in progress.

The supporting systems reviewed during this inspection included the EDG fuel oil, air start, and combustion air systems, and portions of the RHR service water (RHRSW) and ESW systems that supply cooling water to EDGs. The team reviewed the design basis of the selected systems, and modifications to the systems. For the EDG fuel oil and air systems, the team reviewed the volume of fuel oil day tank, the available quantity of fuel, the source of starting air, the air receivers, and associate piping, filters, and supports to verify the support system readiness. The inspectors performed walkdowns of the support systems to verify that the system configuration was consistent with the design basis, and the systems and associated components appeared capable to support intended safety related functions. In addition, the team reviewed relevant procedures and analysis to determine the technical adequacy and clarity of the procedures.

The team also reviewed the application of Nondestructive Examination (NDE) to track degradation of ESW piping due to erosion/corrosion. This included a review of the RHRSW-ESW piping integrity (minimum wall) program requirements, and the adequacy of NDE test procedures, trend analysis of acquired data, data acquisition and evaluation.
The team also reviewed the water clean-up program and procedure to maintain suppression pool cleanliness. This included foreign material control, physical and chemical composition and source term of sludge, the rate of sludge accumulation, and the continuing validity of original assumptions. The inspectors evaluated the effectiveness of the these activities to assure adequate suppression pool water cleanliness such that the RHR in-take strainers would not clog during normal and emergency operation.

The team reviewed the procedure and process for implementing temporary modifications, especially, the technical validity and engineering rationale for the temporary modifications that were continued beyond ninety days; the adequacy and implementation of procedures controlling acquisition, review, and use of vendor/supplier manuals; and technical/maintenance requirements and their incorporation in plant procedures. This review also included the technical adequacy and effectiveness of the temporary shielding program in the plant for long term application of lead shielding blankets, changes in dead weight of piping in the original design and analysis inputs, and the safety evaluations (10 CFR 50.59) requirements, especially for the RHR system piping.

b. Findings

.1 RHR “Soft Fill” Methodology (Suppression Pool Cooling Mode)

Introduction: The team identified a finding regarding inadequate procedural guidance for placing RHR suppression pool cooling in service during a condition of low RHR loop pressure. The team determined that the existing method contained in the suppression pool cooling procedure lacked sufficient guidance to ensure a significant water hammer would not occur and challenge the integrity of system components during restart of the system after a loss of offsite power (LOOP). This issue was determined to be of very low safety significance (Green) and a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures and Drawings.

Description: The team noted that the UFSAR described the use of the suppression pool cooling mode for periods during normal plant operation. The design of the RHR system was such that if a LOOP transient or a seismically induced LOOP occurred while in the suppression pool cooling mode, the RHR pump would trip resulting in a partial drain down of the system. This would occur because the suppression pool cooling return valves (HV-151F028A/B and HV-151F024A/B) would remain open during pump coastdown. The result would be that the LPCI, RHR Head Spray, and Drywell Spray lines would become void of water from the closed isolation valves in these lines down to the main RHR supply header. Additionally, the LOOP would result in the loss of the condensate transfer pumps which normally supply keepfill pressure to the RHR loop.

During a LOOP scenario with the main steam isolation valves (MSIVs) closed, the suppression pool temperature would increase as the safety relief valves would be actuated to control reactor pressure following the MSIV isolation. This would result in the emergency operating procedures directing the operators to maximize suppression pool cooling. The team noted that section 3.1.8 of operating procedure OP-149-005, revision 20, “RHR Suppression Pool Cooling,” contained guidance to place a depressurized RHR loop in service by performing a “soft fill” of the partially drained RHR loop. The procedure directs the operators to close the heat exchanger inlet and bypass valve and then start an RHR pump on minimum flow. The operators would then throttle open the RHR heat exchanger bypass valve HV-151F048A(B), prior to fully opening up the RHR heat exchanger inlet valve HV-151F047A(B). The team questioned the appropriateness of throttling open the bypass valve and then fully opening up the heat exchanger inlet valve during a condition of a partially voided system, which would result in water being rapidly introduced into a closed system, collapsing the void. Additionally, the team noted that this “soft fill” method was different than what was described in other documented analyses such as EC-VALV-1041, “Generic Letter 95-07 Susceptibility Evaluation, HV-151-F047A(B),” which described slowly cracking open the bypass valve and monitoring pressure prior to fully opening the inlet valve.

During the inspection, PPL initiated condition report (CR) 472602 and determined that insufficient design analysis existed to support the procedural guidance and ensure that a damaging water hammer event would not occur. PPL evaluated the issue and performed a rigorous analyses, EC-049-1065, revision 0, “Technical Basis For A Soft Fill Of RHR Suppression Pool Cooling Following A LOOP,” which supported a revision to the procedure. A revised method was developed which involved a fill process into an open piping system thereby avoiding water hammer concerns.

Analysis: The performance deficiency was that instructions existed within the RHR suppression pool cooling procedure which were not appropriate to the circumstances as required by 10 CFR Part 50 Appendix B, Criterion V. Specifically, a “soft fill” procedural method had been developed which would have unnecessarily challenged the integrity of the RHR system if it had been implemented, by creating the potential for a water hammer condition. The issue was determined to be greater than minor because it was associated with the procedure quality attribute for the mitigating systems and barrier integrity cornerstones. The deficient procedure affected the objective of ensuring the capability and availability of the RHR containment heat removal safety function.

The finding was determined to be of very low safety significance, based on an SDP Phase 3 analysis that estimated a change in core damage frequency (CDF) in the magnitude of E-8 per year of reactor operation. An SDP Phase 3 analysis was used because the performance deficiency and other assumptions required a departure from the Phase 2 process. This bounding analysis used the Phase 2 format for a LOOP initiating event assuming a complete loss of an RHR train that was in suppression pool cooling at the time of the LOOP, due to the “soft fill” procedure performance deficiency. For the two affected LOOP sequences, the initiating event frequency was decreased by two orders of magnitude (from E-2 to E-4), to account for the nominal 1 percent of the time the RHR was run in suppression pool cooling over a year, and the probability of containment heat removal failure was increased by one order of magnitude (from E-3 to E-2) because of the loss of redundancy in the RHR trains for suppression pool cooling. The two LOOP core damage sequences included successes of reactor protection, emergency diesel generators, and high pressure injection systems, high pressure coolant injection (HPCI) or reactor core isolation cooling (RCIC). The remaining mitigation capability included: the unaffected RHR train for containment heat removal and the late water injection (from the firewater or the control rod drive systems) or containment venting.

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, Instructions, Procedures and Drawings, requires that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances. Contrary to the above, operational procedure, OP-149(249)-005, contained steps which were not appropriate to the circumstance in that the actions directed by the procedure for a low system pressure condition (50 psig) could have created a water hammer which would have challenged system integrity. However, because this issue was of very low safety significance (Green) and was entered into the corrective action program as CR 472602, this violation was treated as an NCV, consistent with Section V.I.A of the NRC Enforcement Policy: (**NCV 50-387/388-2003-006-01)**, Inadequate RHR Soft Fill Procedure.

.2 Residual Heat Removal "Hard Card" for Suppression Pool Cooling

Introduction: The team identified a finding regarding inadequate procedural guidance for operation of RHR in the suppression pool cooling mode with a low pressure coolant injection (LPCI) signal present. The team determined that the steps found within the hard card instruction did not establish the proper flowpath for the condition where an RHR pump was placed in service with the condensate transfer system keepfill pressure unavailable. This issue was determined to be of very low safety significance (Green) and a non-cited violation of 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures and Drawings.

Description: The team noted that the Operations Procedure Program was defined by procedure OP-AD-055, revision 0. This procedure defined a hard card as a procedure attachment that summarizes the detailed steps contained in a procedure section. The hard card can be separate from the procedure and used in lieu of the detailed steps in the procedure section. Controlled hard cards were located at key locations in the plant. Discussions with operations personnel indicated the hard cards would be used to implement direction given within EOPs to accomplish uncomplicated tasks in an efficient manner. Hard cards were controlled as an attachment to the procedure that they were associated with. The team noted that the hard cards were limited to relatively simple, routine tasks that did not require extensive prerequisites or cautions and required step by step conditional adherence in accordance with step 9.2.6 of OP-AD-055.

The team determined that the hard card associated with the operation of RHR in the suppression pool cooling mode with a LPCI signal present, Attachment B, of OP-149-005, revision 20, did not properly establish the correct flowpath for suppression pool cooling for the condition where condensate transfer system pressure was unavailable. Specifically, step 3.g. gave the direction to ensure at least one RHR pump was in service or perform several steps to establish the suppression pool cooling function. If RHR loop pressure is less than 50 psig, the hard card directs the operator to close both the RHR heat exchanger inlet and outlet valves prior to starting the pump. The team determined that performing steps 3.g. (2), (3), 3.h and 3.i, as written would result in not establishing a flowpath through the test return valve, HV-151-F024A(B), back to the suppression pool. This was due to the RHR heat exchanger inlet and bypass valves inadvertently ending up in the closed position because of an improper sequence of steps within the hard card.

While the hard card was deficient, the team determined that the same concern was not applicable to the detailed section of the suppression pool cooling procedure because step 3.2.4.(g) would restore a pump in accordance with OP-149-001 section 3.9.6 and establish the proper flowpath. PPL initiated CR 473467 to address the deficiency with the hard card. This concern was applicable for both Unit 1 and Unit 2.

Analysis: The performance deficiency was that instructions existed within the RHR suppression pool cooling hard card procedure which were not appropriate to the circumstances as required by 10 CFR Part 50 Appendix B, Criterion V. Specifically, the hard card for suppression pool cooling with a LPCI signal present had not been properly maintained to ensure that it would establish a proper flowpath under all conditions. The issue was determined to be greater than minor because it was associated with the procedure quality attribute for the mitigating systems and barrier integrity cornerstones. The inadequate procedure affected the objective of ensuring the capability of the RHR containment heat removal safety function due to the improper valve alignment.

The RHR containment heat removal safety function affected was associated with the mitigating systems and barrier integrity cornerstones. The finding was determined to be of very low safety significance based on a bounding SDP Phase 3 analysis using the Standardized Plant Analysis Risk Model (SPAR) revision 3.01 for Susquehanna 1&2 dated December 17, 2002. The analysis produced a change in CDF on the order of 7.6E-8 per reactor year of operation. An SDP Phase 3 analysis was used because the performance deficiency resulted in a change to a human error probability, which required a departure from the Phase 2 process. The bounding analysis changed the chance of the operators failing to properly align the RHR system in suppression pool cooling to once in 400 attempts from the originally assumed value of once in 2000 attempts. This increased chance of operator failure was based on the poor hard card procedure. However, the operators had available the correct steps within the body of the operating procedure, and had training and experience with placing the system in this mode of operation. The condition was conservative and bounding because it did not take into account the chance that a LPCI signal may not be present at the time SPC was initiated. The dominate core damage sequences related to loss of service water (LOSW) and LOOP initiating events. The dominate LOSW sequence included successes of the reactor protection, reactor core isolation cooling, manual depressurization of the reactor coolant system, and containment spray and venting. The remaining mitigation capability included late water injection from the firewater system. The dominate LOOP sequence was similar to the dominant LOSW sequence except the emergency diesel generators were an additional success and the control rod drive system was another remaining late water injection source.

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, Instructions, Procedures and Drawings, requires that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances. Contrary to the above, the hard card, Attachment B, associated with operational procedure, OP-149(249)-005, contained steps which were not appropriate to the circumstance in that the actions directed by the hard card for a low system pressure condition (50 psig) would have resulted in an incorrect flowpath alignment for suppression pool cooling. However, because this issue was of very low safety significance (Green) and was entered into the corrective action program as CR 473467, this violation was treated as an NCV, consistent with Section V.I.A of the NRC Enforcement Policy: **(NCV 50-387/388-2003-006-02)**, Inadequate RHR Suppression Pool Cooling Procedure With A LPCI signal present.

4OA2 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

The team assessed whether licensee personnel were identifying issues with the RHR, EDG, and supporting systems at the proper threshold and entering them in the corrective action program.

Specifically, the inspectors reviewed a selection of Action Requests (ARs), Condition Reports (CRs), Corrective Actions (CAs), self-assessments, and Quality Assurance (QA) audits to verify that problems were identified, documented, and effectively resolved in a timely manner.

b. Findings

No findings of significance were identified

4OA6 Meetings, Including Exit

.1 Management Meeting

The team presented the inspection results to Mr. Bryce Shriver and other members of the PPL staff at an exit meeting on May 16, 2003. The team reviewed some proprietary information during the inspection. This material was either returned to PPL personnel or destroyed. The team verified that this inspection report does not contain proprietary information.

ATTACHMENT 1

SUPPLEMENTARY INFORMATION

Key Points Of Contact

Licensee Personnel

R. Bogar	Station Engineering
P. Brady	Electrical/I&C Design Supervisor
K. Browning	Manager - Nuclear Support, Mechanical Design
S. Ellis	Station Engineering - Electrical/I&C Supervisor
D. Gladey	Senior Engineer/Electrical Design
G. Jones	Vice President - Special Projects
T. Kirwin	Maintenance Manager
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R. Peal	Assistant Operations Manager
K. Roush	Manager - Nuclear Training
R. Saccone	General Manager - Nuclear Engineering
J. Schleicher	Supervisor - Design Engineering
R. Sgarro	Manager - Nuclear Regulatory Affairs
J. Shaw	Manager - Station Engineering
B. Shriver	Senior Vice President / Chief Nuclear Officer
J. Tripoli	Senior Engineer - Nuclear Regulatory Affairs
A. White	Senior Engineer - Electrical/I&C
A. Wrape	General Manager - Nuclear Assurance

NRC Personnel

L. Doerflein	Systems Branch Chief
S. Hansell	Senior Resident Inspector
W. Lanning	Director, Division of Reactor Safety

List of Items Opened, Closed, and Discussed

Opened

None

Opened and Closed

50-387/388-2003-006-01	NCV	RHR Soft Fill after LOOP
50-387/388-2003-006-02	NCV	RHR Hard Card vs procedural difference

List of Acronyms

AR	Action Requests
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CR	Condition Report
CA	Corrective Actions
DBD	Design Basis Document
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
NCV	Non Cited Violation
NDE	Nondestructive Examination
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
P&IDs	Piping & Instrumentation Drawings
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
SDP	Significance Determination Process
SPC	Suppression Pool Cooling
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
VDC	Volts - Direct Current

List of Documents Reviewed

Design and Licensing Basis Documents

DBD-014, Rev 2, Design Basis Document for Residual Heat Removal System
DBD for the EDGs, for the following chapters:
- 2.5, Fuel Oil Functional Requirements
- 2.6, Air Start System Functional Requirements
- 2.8, Combustion Air System Functional Requirement
- 2.4.1.5.2, Combustible materials
- Chapter 1, Appendix A, Open Item list for Diesel Generators and Auxiliaries

SSES - UFSAR, Sections:
- 3.1.2.5.6, Reactor Coolant Pressure Boundary Penetrating Containment
- 5.4.7, Residual Heat Removal System
- 6.2, Containment Systems
- 6.3, Emergency Core Cooling Systems
- 6.3.2.2.5, Discharge Line Fill System
- 9.5.4, Diesel Generator Fuel Oil Storage and Transfer system
- 9.5.6, Diesel Generator Starting Air System
- 9.5.8, Diesel Generator Combustion Air Intake and Exhaust System
- 17.2, Quality Assurance During the Operations Phase

Technical Specifications:
- 5.5.9, Diesel Fuel Oil Testing Program

3.8.3, Diesel Fuel Oil, Lube Oil and Starting Air, and testing
3.4, Reactor Coolant System
3.5, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling
3.6, Containment Systems
Unit 1 Technical Specification Amendment 178

GE Product Safety Standard 22A8412, Rev 0 (RHR System)
GE Product Safety Standard 22A8409, Rev 1 (ECCS System)
Bechtel Design Specification 8856-M-170
General Bechtel Specification 8856-M-406

Procedures and Surveillance Tests:

SC-023-002, New Diesel Fuel Oil Receipt Analyses
SC-023-003, Rev. 2, 31 Day Particulate Analysis and Water Check on the 'A' EDGS
SE-023-003, Rev. 2, Diesel Fuel Oil Transfer System Flow Verification
SE-124-107, Rev. 12, Unit 1, Div. 1 Diesel Generator LOCA/LOOP Test Data
SE-124-207, Rev. 13, Unit 1, Div. 2 Diesel Generator LOCA/LOOP Test Data
SE-224-107, Rev. 12, Unit 2, Div. 1 Diesel Generator LOCA/LOOP Test Data
SE-224-207, Rev. 13, Unit 2, Div. 2 Diesel Generator LOCA/LOOP Test Data
MT-RC-004-1, Rev. 5, Relay Calibration Data Form
SM-202-A04, Battery Performance Discharge Test
Z-0006-51, 2DG10 Performance Test Data March 2003
Z-0007-51, 2DG20 Performance Test Data March 2003
Z-0008-51, 2DG30 Performance Test Data March 2003
OP-024-001, Rev. 39, Diesel Generator Operations Procedure
OP-024-004, Rev. 23, Transfer and Test Mode Operations of DG 'E'
SO-054-A03, Rev. 3, Quarterly ESW Flow Verification LOOP A
SO-150-004, Rev. 20, Quarterly RCIC Valve Exercising
SM-204-001, Rev. 14, 4KV Bus 24 month UV Channel Calibration
SM-024-002, Rev. 12, 24 Month Emergency Diesel Engine Inspection
SM-202-403, Rev. 15, Service Discharge Test
NDAP-QA-0401, Rev. 3, EDGS Reliability Monitoring Program
OI-024-004, Rev. 13, DG Operating Log Form
MT-024-028, Rev. 5, Predictive Maint. Monitoring Program
MT-RC-012, Rev. 10, SA-1 Relay Calibration
MT-AD-605, Rev. 7, Maint. and Calibration of Installed Plant Instrumentation
MT-EM-001, Rev. 7, Relay Calibration Activities
SO-24-005, Rev. 0, DG Air Start Operability Data Form
SO-024-A02, Rev. 0, DG 'A' Air Start Receiver Check Valve Test
NDAP-QA-0103, Rev. 7, Audit and Independent Assessment Programs
NASP-QA-0401, Rev. 7, Internal Audits
NDAP-00-0745, Rev. 2, Self-Assessment and Benchmarking Program
NDAP-QA-0409, Rev. 3, Door, floor plug, and hatch control
NDAP-QA-0702, Rev. 12, Action request and Condition report process
OESI-AD-001, Rev. 4, Action request process – subtype condition report (CR) and management (MGNT) processing"
H-1019, Rev. 2, Inspection Program for Piping Corrosion and Degradation Index
NDAP-QA-1218, Rev. 2, Temporary Modifications
NDAP-QA-1202, Rev. 7, Nuclear Department Modification Program

ME-059-001, Rev. 2, Suppression Pool Cleaning, Inspection, and Underwater Work
NDAP-QA-1213, Rev. 2, Control and Use of Vendor Technical Information
NDAP-QA-0404, Rev. 4, Att. E, Shadow Shielding Design Requirements
SO-149-B02, Rev. 9, Quarterly RHR System Flow Verification Div II
SO-149-B05, Rev. 6, Quarterly RHR LOOP B Valve Exercising
SO-149-001, Rev. 13, RHR Monthly Alignment Check
SE-149-002, Rev. 12, 24 Month RHR System Logic Functional Test Div II
SM-249-001, Rev. 2, 24 Month Calibration Check of RHR Pump
OP-149-001, Rev. 25, RHR System
OP-149-004, Rev. 17, RHR Containment Spray
OP-249-002, Rev. 36, OP-149, Rev. 33, RHR Shutdown Cooling
OP-149-005, Rev 20, RHR Suppression Pool Cooling
OP-149-005, Attachment B, RHR Suppression Pool Cooling Mode With LPCI
EO-000-102, Rev. 1, RPV Control
SSES-EPG, Rev. 8, SSES-EPG/SAG, PRV Control Guidelines, Purpose
EO-000-103, Rev. 2, Primary Containment Control
EO-000-113, Rev. 1, Level/Power Control
SO-249-014 Rev. 10, RHR Cold Shutdown Valve Exercising
SO-249-A05, Rev. 7, Quarterly RHR Loop A Valve Exercising
SO-292-B02, Rev. 7, Quarterly RHR System Verification Div II (SPS SO-249-002)
ON-104-001, Rev. 13, Unit 1 Response To Loss Of All Offsite Power
ON-037-001, Rev. 3, Loss Of Condensate Transfer System
OP-AD-055, Rev. 0, Operations Procedure Program
SSES-EOP Revision 8 Contingency 1, Alternate Level Control
SSES-EOP, Revision 9, Caution #6
SO-150-002, Quarterly RCIC flow verification

<u>Drawings/ Change notices:</u>

F124510 Sheets 301 and 302, Rev. 11,8, Process Diagram, Residual Heat Removal System
E106256 Sheets 1-5 Rev. 55,47,18,14,1, Susquehanna SES Unit 1, P&ID Residual Heat Removal
E-106178 Sheets 1- 3, Plant Design Drawing, Reactor Building Unit 1, Area 28,
E106179 Sheets 1- 3 Plant Design Drawing, Reactor Building Unit 1, Area 29
FF123410 Sheets 301 and 302 , Rev. 11 and Rev. 8
GBB-105, Rev. 6, Isometric, Reactor Building
HBB-110-1- 4, Isometric, Reactor Building, RHR, Unit 1
E105951 Sheets 1-5, Susquehanna SES Unit 2 P&ID RHR
Interim Drawing Change Notice, Modification Package No. DCP 94-94-9081 Rev. 6F13, Pages 1-5
E-11 Series, 125 and 250 VDC Single Line Diagrams
E-153 Series, Schematic Diagrams
E-23-1, Rev. 26, 4KV Meter and Relay Diagram
E-103 Series, 4KV Bus Relay Diagrams
E-105 Series, 4KV Diesel Generator Schematics
E-184 Series, DG Auto Start Signals
E-259 Series, DG and Auxiliary Control Schematics
M1-E11-11, Rev. 3, RHR Pump Motor Speed Torque
M1-E11-12, Rev. 6, RHR Pump Outline
M1-E11-29/66, Rev. 26/15, RHR System Elementary Diagrams

M30-124 Series, DG Control Schematics

Calculations:

EC-002-0506, Rev. 17, 125 VDC Unit 1 Battery Load Profiles
EC-002-1031, Rev. 3, Load Profile for Modified Performance Test
EC-004-0502, Rev. 4, Degraded Grid UV Relay Setpoints
EC-004-0503, Rev. 0, Degraded Grid Scheme Tolerance Calculation
EC-004-0537, Rev. 0, Design Basis for Degraded Grid Protection
EC-004-1002, Rev. 6, LOCA Time Line Development for Voltage Studies
EC-004-1010, Rev. 2, Degraded Grid Voltage Study
EC-024-0503, Rev. 14, Diesel Generator Load Calculation
EC-049-0001, Rev. 4 Pressure Drops in RHR System for Various Modes of Operation
EC-049-0009, Rev. 0, RHR Pump Suction and Discharge Line Pressure
EC-049-0509, Rev. 0, Water Hammer Project Pressure Drop for Suppression Pool Cooling
EC-049-512, Rev. 1, RHR Injection Stroke Time
EC-049-0650, Rev. 0, Reactive Thrust Force for RHR Suction Side Relief Valves PSV 151F030A/B/C/D and PSV 251F030 A/B/C/D
EC-049-0669, Rev. 0, Pressure Locking of RHR and Core Spray
EC-049-0691, Rev. 1, Calculation of RHR & Core Spray Pump NPSH Limits
EC-059-0012, Rev. 2, Suppression Pool Suction Strainer Pressure Loss
EC-059-0534, Rev. 0, Suppression Pool Response to LOCA events
EC-059-1002, Rev. 0, Realistic NPSHA for Core Spray and RHR Pumps
EC-059-1032, Rev. 1, CS and RHR Stacked Disk Strainer Stress Analysis
EC-059-1034, Rev. 0, Test Report and Sizing Calculation for Core Spray and RHR Stacked Disk Suction Strainers
EC-059-1036, Rev. 1, Basis for ECCS & RCIC FSAR Net Positive Suction Head (NPSH)
EC-EOPC-0502, Rev. 2, Calculation of Work Sheet #13 Parameters (Vortex Limits for ECCS Pumps) for BWR Owners Group Emergency Procedure Guideline
EC PUPC 0519, Rev. 1, Evaluation of piping system for uprated conditions
EC-PUPC-1001, Rev. 5, GE Power Uprate Engineering Report
EC-PIPE-16041, Rev. 2, PSTR, Evaluation of GE Replacement Strainers for RHR and Core Spray Pump Suction Piping
EC-SOPC-0503, Rev. 1, Relay Setting Calculation for RHR pump
EC-SOPC-0514, Rev. 0, Relay Setting Calculation for Degraded Grid Protection
EC-SOPC-0605, Rev. 0, Relay Setting Calculation for ESW pump
EC-THYD-1034, Rev. 1, Submerged Drag Loads, Replacement ECCS Suction Strainers
EC-049-0654, Rev. 0, Justification for RHR HX cooling water flow
EC-049-0513, Rev. 0, RHR Draindown Test Results
EC-049-0648, Rev. 1, Loss of ECCS Keepfill Evaluation of App. R Manual Actions
EC-049-0649, Rev. 0, Alternate RHR fill method during loss of condensate transfer
EC-VALV-1041, Rev. 4, GL 95-07 HV-151-F047A(B)
EC-080-1009, Rev. 0, Reactor Water Level Setpoints
EC-058-1012, Rev. 1, Primary Containment High Pressure Setpoint

QA Audits

2003-2004 QA Internal Audit Schedule
2001-010, Corrective Action Program Audit

2002-033, Modification Program Audit
2002-054, Configuration Management and Replacement Item Evaluation Audit
2002-030, Maintenance / Repair Program

Self-Assessments

2002 Self-Assessment Plan Query
MOD-01-08, TMODS Process
MOD-02-05, Modifications Overall U1-12RIO
NSE-02-022, Use-As-Is Dispositions
NSE-02-05, Post-Maintenance/Modification testing
NSE-02-04, MR Documentation Consistency SE's
NCM-02-05, Program Compliance of Configuration Management in the Modification Issuance Process
MOD-02-17, Modification Process Benchmarking
Weekly CAP Status Report, April 30, 2003

Action Requests (AR) and Condition Reports (CR)

AR 381523	AR 449281	AR 461454	CR 413047	CR 449281	CR 471120*
AR 383170	AR 449293	AR 466945	CR 417587	CR 454414	CR 472285*
AR 395049	AR 449297	CR 198084	CR 417914	CR 454959	CR 472404*
AR 418909	AR 452911	CR 276714	CR 419835	CR 455822	CR 472602*
AR 442694	AR 452130	CR 291582	CR 429831	CR 459719	CR 473467*
AR 442944	AR 452482	CR 357969	CR 434206	CR 461850	CR 473654*
AR 443846	AR 453237	CR 358187	CR 435654	CR 465947	CR 473727*
AR 445095	AR 454963	CR 358380	CR 436068	CR 468877	CR 473769*
AR 447710	AR 455543	CR 359015	CR 437402	CR 470098*	CR 474080*
AR 444828	AR 458564	CR 363590	CR 445170	CR 470104*	CR 55883
AR 444907	AR 458959	CR 363753	CR 445361	CR 470935*	CR 70952
AR 448848	AR 460227	CR 365876	CR 446537	CR 470949*	CR 87360
AR 448897	AR 461402	CR 405313	CR 446854	CR 470978	CR 92347

(Note " * " = Generated as result of inspection)

System Health Reports

Units 1, 2, & 0, Station Health, Third Quarter 2002 (systems relevant to inspection scope)
24-Diesel Generators, First Quarter 2003
Unit 1, 116-RHRSW RHR Service Water, First Quarter 2003
Unit 2, 149, RHR Residual Heat Removal, First Quarter 2003
Unit 2, 216-RHRSW RHR Service Water, First Quarter 2003
Unit 2, 249-RHR Residual Heat Removal, First Quarter 2003
Station Status Report, April 28, 2003

Temporary Modifications:

NIMS Report ID: WMXR166, 5-13-2003, Installed TMOD'S sorted by unit with associated W.O Records, Open TMODs and list of TMODs completed in 2002-2003

NIMS 325842, Temporarily Remove LSL-01707 from Service
TSR No. 02-0006, Temporary Shielding Request data sheet, 1-25-2002
Form NDAP-QA-1218-1, Rev.0, Temporary Modification cover sheet
TSR No 97-0003 & 0004 &0005, Temporary Shielding Request Data Sheet, May 1, 1997
TSR No 98-0026, Temporary Shielding Request Data Sheet, October 13, 1998
TSR No 02-0006, Temporary Shielding Request Data Sheet, January 25, 2002
NDAP-QA-0404, 50.59 and 72.48 Screening Determination, February 22, 2000

ECCS strainers:

NRC Bulletin 96-03: Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water-Reactors, May 6, 1996
EC-059-1012, Evolution of Debris Effects on ECCS
EC-CHEM-1001, Rev.4, Suppression Pool Sludge Quantification
IERP No 98077, Potential for Degradation of ECCS
IERP No 95139, Unexpected Clogging of a RHR Pump strainer While Operating in Suppression Pool Cooling

Work Orders:

Work Order: PCVO 247496, Replace Motor and verify status of MOD. 93-9063
Work Order: PCWO 324642, HV151F016A, Inspect Motor pinion gear setscrew

Miscellaneous

GEK-71400, General Electric Document, Susquehanna Emergency Core Cooling Systems, Volume VII, Part 2
PLA 1264, SSES Response to DG Air Start Operations, September 1982
NEIM-AD-0213, Rev 4, Design Base Document Open Items Management
NUREG/CR-2772, ARL-398A, Sand82-7064, Hydraulic Performance of Pump Suction Inlets for Emergency Core Cooling Systems in Boiling Water Reactors
NEDO-31331, Class 1, March 1987, BWR Owners Group Emergency Procedure Guidelines Revision 4
Safety/Relief Test Records for RHR Pump A, B, C, D Suction Relief Valves
Component Data Sheets for Units 1 and 2 LPCI Injection Loop A and B
Component Data Sheets for RHR Pump A, B, C, D Suction Relief Valve
Replacement Item Evaluation RIE 91-0150 Rev. 4, Replacement of PSV 1/251 F025 A/B
Relay Calibration Data Forms, 1989-2003
006-36049 RHR Pump Performance Curves, May 18, 1976
Risk Informed Inspection Notebook for Susquehanna Steam Electric Station (SSES) Units 1 and 2, Rev. 1
CDI Technical Note No 98-03 Rev. 1, Data Report for PPL Core Spray and RHR ECCS Suction Strainers Vortex and Clean Head Loss Tests.
Summary of recurrent AR's and CR's (Years 2002 & 03)
Susquehanna Steam Electric Station response to NRC request for information regarding adequacy and availability of Design Basis information, PLA-4546, File R41-2, dated February 13, 1997

Design Basis Documentation Program, Summary sheets of Open Items, including “Resolution to
Open Item DBD013.040"

Surveillances:

Surveillance and Analyses of new diesel fuel oil receipt: NDAP-QA-0722-1, preformed March 21, 2003, and SC-023-002-1, Rev.1, performed between March 19 and 21, 2003
31 Day particulate and water check of the fuel oil storage tank of “A” D/G: SC-023-003, Z2155-01, and NDAP-QA-0722-2, Rev. 4, performed between March 25 and 26, 2003
Surveillance of the “E” D/G fuel oil transfer system: SE-023-003, Z2033-01, and SE-023- 003, and REWL-41436, performed between August 14, 2002 and October 2, 2002
Quarterly ESW Flow Verification LOOP A: SO-054-A03, and SO-054-A03, Rev. 3, performed April 9, 2003.